

**U.S. NUCLEAR REGULATORY COMMISSION
REGION I**

License No.: DPR-35

Report No.: 98-04

Docket No.: 50-293

Licensee: Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Facility: Pilgrim Nuclear Power Station

Inspection Period: April 27, 1998 - May 1, 1998 (on-site)
May 1, 1998 - May 15, 1998 (in-office)

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EXECUTIVE SUMMARY

Pilgrim Nuclear Power Plant Inspection Report No. 50-293-04

This inspection reviewed Pilgrim's implementation of 10 CFR 50.65, the maintenance rule (MR).

MAINTENANCE

- Systems, structures and components (SSCs) performance criteria for reliability and unavailability were conservatively established for most systems and were directly related to the failure rates assumed in the probabilistic risk assessment (PRA). However, Boston Edison Company's (BECo's) failure to consider the anticipated transient without scram mitigation function and establish appropriate performance criteria for the control rod drive system was a violation of 10 CFR 50.65 (a)(2). (section M1.1)
- The condition monitoring program for structures and the overall material condition of the SSCs walked down were good. (section M1.1)
- Corrective actions were taken when a SSC failed to meet its goal, performance criteria, or experienced a functional failure with some exceptions. BECo permitted the primary containment and feedwater systems to remain under 10 CFR 50.65(a)(2) when preventative maintenance failed to assure that these SSCs remained capable of performing their intended function in violation of 10 CFR 50.65(a)(2). (section M1.1)
- BECo's SSC scoping, SSC function identification, and system boundary descriptions were generally acceptable. However, in violation of 10 CFR 50.65(b) BECo failed to include in the scope of the rule the heating, ventilation and air conditioning system for the reactor building 480V switch gear and the firewater system backup supply to the screenwash system. (section M1.2)

In addition, BECo added 12 SSCs to the MR scope after the required implementation date of July 10, 1996. BECo is being credited for identifying this aspect. (section M1.2)

- The methods and calculations that BECo established for making risk determinations, and for establishing performance criteria were acceptable. The expert panel's decisions regarding the performance criteria, risk ranking and knowledge of on-line and shutdown maintenance risk assessment were appropriate. (section M1.3)

- BECo's program adequately balanced availability and reliability. However, BECo's (a)(3) assessment should have been completed at the end of RFO 11 in April 1997 and was not completed until after the inspection in violation of 10 CFR 50.65 (a)(3). (section M1.4)
- The process for assessing risk associated with scheduled maintenance work activities was generally good and was being properly implemented at the site. Probabilistic risk assessments were not used to assess the overall plant risk in certain conditions. The persons interviewed expressed sufficient knowledge of the risk assessment process to implement the maintenance rule program. BECo's approach to shutdown risk program was also reasonable. (section M1.5)
- System engineer and operations department personnel knowledge of the MR and their associated responsibilities was adequate to ensure acceptable implementation of the maintenance rule. (section M.3)
- Program revision and substantial improvements were implemented just prior to the inspection due, in part, to the thoroughness of the licensee's self-assessment and audit processes (section M.7)

Report Details

M1 Conduct of Maintenance (62706)

M1.1 Goal Setting and Monitoring (a)(1), Preventive Maintenance (a)(2)

a. Inspection Scope

The team reviewed Boston Edison Company's (BECo) program documents in order to evaluate the process established to set goals and monitor under (a)(1) and to verify that preventive maintenance had been demonstrated to be effective for systems, structures and components (SSCs) under (a)(2) of the maintenance rule. The in-depth vertical slice assessment on each SSC included a verification that goals and performance criteria were established in accordance with safety, industry-wide operation experience was taken into consideration, appropriate monitoring and trending were being performed, and that corrective actions were taken when a SSC failed to meet its goal, performance criteria, or experienced a functional failure (FF). The team also discussed the program and performed a system walkdown to assess SSC material condition with the responsible system engineer. Deep vertical slice assessments were performed on the following SSCs:

- MSIV, (a)(1)
- Primary containment, (a)(1)
- Instrument air, (a)(1)
- 23KV, (a)(1)
- RPIS, (a)(1)
- Reactor manual control, (a)(1)
- Screen wash, (a)(1)
- Feed water/condensate, (a)(1)
- PASS, (a)(2)
- Control room high efficiency air filtration system, (a)(2)
- Standby gas treatment system, (a)(2)
- Control rod drive, (a)(2)
- Intake structure, (a)(2)
- Spent fuel pool cooling, (a)(2)

b. Observations and Findings

The team reviewed a sensitivity study that the licensee had performed for the high risk significant systems which input all of the availability criteria and also considered the plant level criteria as the initiating events frequencies in the probabilistic risk assessment (PRA). Based on this scenario, the core damage frequency (CDF) decreased from 2.8 E-05 to 2.6 E-05 per reactor year. The PRA was re-evaluated during the inspection using the original initiating events frequencies which had been input into the 1995 PRA update. This re-evaluation determined that the CDF increased by approximately 15% to 3.2 E-05 per reactor year (based on a truncation limit of 1E-09). The team considered this to be a reasonable increase in CDF.

The FF limits were tied to the PRA assumed failure rates. The demand failure probabilities were multiplied by the number of demands experienced by the particular system in a two year period. The running failure rates were multiplied by the number of running hours experienced by the particular system in a two year period. [The team noted that this method results generally in more restrictive performance criteria than would result from application of the binomial theorem (Bernoulli process) for standby systems and the Poisson distribution for operating systems.] The sum of the two failure probabilities determined the reliability performance criterion for that particular system. The team reviewed a sensitivity study that the licensee had performed that factored the reliability performance criteria into the sensitivity calculation performed for the availability criteria, using the 1995 individual plant examination (IPE) initiating events frequencies. The result considering both the availability and reliability performance criteria together was a CDF of 5.6 E-05 per reactor year. The team considered this to be a minor increase in CDF.

The team concluded that reliability and availability performance criteria were appropriately established based on both the PRA data and actual historical performance. In addition, the performance criteria and goal setting for all systems reviewed were adequate with the exception of the Control Rod Drive (CRD), feed water (FW), and primary containment systems. The team identified that BECo failed to adhere to the following 10 CFR 50.65 requirements as follows;

- (1) Control Rod Drive (CRD) - The team identified that the risk significant anticipated transient without scram (ATWS) mitigation function of the CRD system was not addressed in the basis document. In response to the teams finding, the expert panel determined that the CRD pumps were risk significant for their ATWS mitigation function and therefore, unavailability criterion was required. BECo's failure to establish appropriate performance criteria for the CRD system is an apparent violation of 10 CFR 50.65 (a)(2). (EEI 50-293/98-04-01)
- (2) Feedwater (FW) - BECo's review identified the system should have been placed in an (a)(1) status by May 1997 but the status change had not occurred until December 1997. The status change was required due to repetitive functional failures on the feedwater regulating valves. BECo's failure to place the systems into an (a)(1) status in a timely manner commensurate with safety is an apparent violation of 10 CFR 50.65 (a)(2). (EEI 50-293/98-04-02)

- (3) Primary Containment (PC) - BECo's review identified the system should have been placed in an (a)(1) status by July 10, 1996 but the status change had not occurred until November 1997. The status change was required due to in-service and local leak rate test functional failures that occurred in April 1995 which exceeded the established system performance criteria. BECo's failure to place the systems into an (a)(1) status in a timely manner commensurate with safety was an apparent violation of 10 CFR 50.65 (a)(2). (EEI 50-293/98-04-03)

Although BECo identified violations (2) and (3), the violations resulted apparently because of a past problem (organizational ineffectiveness) identified in October 1997 (see also section M.7). These systems were in a(1) status at the time of the inspection for the correction of problems associated with not meeting a(2) criteria. However, corrective actions to prevent repetition were not specifically addressed in the licensee's internal problem reporting system.

The team reviewed goals and corrective actions established by BECo for identified (a)(1) SSCs and found them to be acceptable. Each of the SSC functional failures reviewed were known and understood by the responsible system engineer (SE), had been suitably captured in the problem report (PR) program, and had appropriate corrective actions instituted. The team also discussed the program and performed a system walkdown to assess SSC material condition with the responsible system engineer. No significant problems were noted.

In addition, the team reviewed the structural monitoring program and determined that this program was good. During plant tours several members of the team inspected selected structures including tanks, supports, seismic wall, and foundations. No discrepancies were noted.

c. Conclusions

SSC performance criteria for reliability and unavailability were conservatively established for most systems and were directly related to the failure rates assumed in the PRA. However, BECo's failure to consider the ATWS mitigation function and establish appropriate performance criteria for the CRD system is an apparent violation of 10 CFR 50.65 (a)(2).

Corrective actions were taken when a SSC failed to meet its goal, performance criteria, or experienced a FF. Additionally, the condition monitoring program for structures and the overall material condition of the SSCs walked down were good.

BECo permitted the primary containment and feedwater systems to remain under 10 CFR 50.65(a)(2) when preventative maintenance failed to assure that these SSCs remained capable of performing their intended function. These are apparent violations of 10 CFR 50.65(a)(2).

M1.2 Structures, Systems, and Components (SSCs) included within the Scope of the Rule

a. Inspection Scope

The team reviewed scoping documentation which included the BECo Updated Safety Analysis Report (UFSAR), Emergency Operating Procedures (EOPs), and individual SSC basis documents which identified system boundaries to determine if the appropriate SSCs were included within BECo's maintenance rule (MR) program.

b. Observations and Findings

BECo Maintenance Rule Procedures NE16.03 and NOP95A4 identified the methodology for selecting SSCs and SSC functions that should be included within the scope of the rule. BECo had identified that 105 of 140 individual SSCs were within scope when the maintenance rule was implemented in July, 1996. The team also verified that adequate technical justification was provided for those SSCs and/or functions excluded from scope.

The team determined that BECo's SSC scoping, SSC function identification, and system boundary descriptions were adequate with the following exceptions;

- BECo added 12 SSCs to the scope of the rule after initial MR implementation on July 10, 1996, which was a violation of 10 CFR 50.65(b). BECo is being credited with self identifying this problem.
- The team identified that the heating, ventilation and air conditioning (HVAC) units for the reactor building 480V switch gear environmental enclosures were not included in the scope of the MR. These HVAC units were designed to protect the associated switchgears (which provide power to essential equipment loads) from high energy line breaks in the secondary containment. These units should have been included in the scope of the MR because the associated switchgears are relied upon to mitigate the consequences of an accident or transient. This is an example of an apparent violation of 10 CFR 50.65 (b). (EEI 50-293/98-04-04)
- The team identified that the firewater system function of providing a backup water supply to the screenwash system was not included in the scope of the rule. It appeared that this function should have been included in the scope of the MR because its failure could result in a scram or safety system actuation. This is an example of an apparent violation of 10 CFR 50.65 (b). (EEI 50-293/98-04-04)

In response to the team's findings, the expert panel met during the inspection regarding the team's findings that the HVAC units for the reactor building 480V switch gear environmental enclosures and the firewater system function of providing a backup water supply to the screenwash system. The team noted that the expert panel took appropriate steps to add these functions to the BECo MR scope and to establish appropriate performance criteria.

c. Conclusions

BECo's SSC scoping, SSC function identification, and system boundary descriptions were generally acceptable. However, BECo's failure to include in the scope of the rule the HVAC for the reactor building 480V switch gear environmental enclosures and the firewater system function of providing a backup water supply to the screenwash system are examples of an apparent violation of 10 CFR 50.65 (b).

Additionally, BECo added 12 SSCs to the MR scope after the required implementation date of July 10, 1996. BECo is being credited with self-identification for these issues.

M1.3 Risk Ranking and Expert Panel

a. Inspection Scope

The team reviewed the methods and calculations that BECo established for making risk determinations, and for the performance criteria established for the specific SSCs evaluated during this inspection. In addition, the team reviewed the BECo's expert panel process, panel meeting minutes and attended an expert panel meeting.

b. Observations and Findings

Risk ranking

The process for determining the risk significance of SSCs within the scope of the Maintenance Rule was documented in Procedure NOP95A4, "NRC Maintenance Rule Procedure," Revision 1 and Procedure 16.03, "10 CFR 50.65 NRC Maintenance Rule," Revision 1. The process was based on the PRA model developed for the individual plant examination (IPE) of severe accident vulnerabilities and was a linked fault tree model which was developed using the EPRI-developed CAFTA code. The calculated core damage frequency (CDF) for the 1992 submittal was 5.8 E-05 /reactor year for internal initiating events. The licensee responded to an NRC Request for Additional Information (RAI) and revised the PRA model and determined a new CDF of 2.8 E-05 /reactor year. However, the data generally were those used in the 1992 IPE which had a cutoff date of September 30, 1989, with the exception of the high pressure coolant

injection (HPCI) and reactor core isolation cooling (RCIC) systems which had a cutoff of March 31, 1992. The team noted that the HPCI and RCIC systems had been singled out (i.e., used more current IPE insights) because of the high frequency of loss of offsite power due to severe weather conditions and consequent isolation of the main steam system and reliance on the HPCI and RCIC systems for safe shutdown.

A total of 85 systems or sub-systems were modeled in the PRA. PRA basic events were matched to their corresponding system design basis function in the plant's final safety analysis report (FSAR). The resulting system functions were ranked by the Fussell-Vesely measure ($F-V > 0.005$), the risk achievement worth measure ($RAW \geq 2.0$), the risk reduction worth measure ($RRW > 1.005$) and the top 90% of CDF cutsets measure. If the SSC satisfied any one of the measures, it was considered to be high risk significant. The systems function ranking was based on the 1995 PRA model. A systems ranking based on the 1992 PRA had been performed using the F-V and RAW measures. During the inspection, a systems ranking based on the 1995 PRA was generated. There were no new systems identified based on the 1995 PRA ranking. As compared to a 1995 baseline CDF of $2.8 \text{ E-05/reactor year}$, the cutsets appearing above a truncation point of 1E-09 were used for the risk ranking process. As noted in the December 28, 1995 response to the RAI, cutsets falling below the truncation point were reviewed to assure that no important components affected by operator recovery actions had been lost by the truncation process.

Each function was reviewed by the expert panel to establish its risk significance. Eighty-two (82) systems or subsystems had maintenance rule functions which were considered risk significant. Of those, fourteen (14) functions were upgraded by the expert panel prior to the start of this inspection. No functions had been downgraded by the expert panel.

Risk significance of components used during shutdown had been determined qualitatively using the outage safety assessment guidelines of NUMARC 91-06. The critical shutdown functions were controlled by Chapter 4, "Planned Outage Safety Reviews" of the station outage management guidelines. The team found this method to be acceptable.

Overall, the risk ranking process was acceptable with some exceptions. Appropriate actions had been taken by the expert panel prior to the inspection regarding the risk ranking process. However, the team identified the potentially high risk significance of the CRD pumps and the HVAC for the 480 vac MCC enclosures (discussed previously in sections M1.1 and M1.2).

Expert Panel

The instructions and responsibilities for the formation and operation of the expert panel were identified in Procedure NOP95A4, "NRC Maintenance Rule Procedure," Revision 1. The procedure specified that members of the Operations, Maintenance and Projects, Nuclear Engineering Services Group, and System and Safety Analyses would be represented. In addition, the Maintenance Rule Coordinator was the chairman of the panel and, at the time of the inspection, was the senior PRA expert at the plant. The team considered the backgrounds and qualifications of the panel members to be appropriate. The training of the members in PRA was adequate for the members to sufficiently understand the concepts of PRA and to perform their responsibilities.

The responsibility for reviewing and approving (a)(1) goals and corrective actions and approving the return of SSCs to (a)(2) status was assigned to a separate panel called the Design Review Board. This panel consisted of senior management and technical personnel whose primary function had always been to review design modifications.

In February 1998, BECo reconstituted the expert panel. The panel had been inactive since January 1997. The team reviewed the meeting minutes issued since February 1998 and determined that the meeting minutes lacked detail and did not adequately identify the discussions and the reasons for the decisions taken. No formal meeting minutes were published before February 1998.

The expert panel approved the risk ranking and performance criteria. The final decisions made by the expert panel were documented by means of the risk ranking and performance criteria in the Maintenance Rule SSC Design Basis Documents.

The team interviewed the operations member of the expert panel. This member's knowledge of both the online and shutdown maintenance risk assessment programs and the plant's procedures to minimize on-line and shutdown risk was appropriate.

Several team members attended a meeting of the expert panel that was convened to review concerns identified by the team. The panel deliberated the status of the HVAC units for the reactor building 480V switch gear environmental enclosures and the ATWS mitigation function of the CRD pumps for risk significance and performance criteria. The panel subsequently concluded that the HVAC units for the reactor building 480V switch gear environmental enclosures should be placed in the MR scope and that the ATWS mitigation function of the CRD pumps should be considered high risk significant and placed in scope. The team considered the deliberations to be thorough and to adequately address the issues involved in implementing the maintenance rule.

c. Conclusions

The methods and calculations that BECo established for making risk determination, and for establishing performance criteria were acceptable. The expert panel's decisions regarding the performance criteria, risk ranking and knowledge of on-line and shutdown maintenance risk assessment were appropriate.

M1.4 (a)(3) Periodic Evaluations and Balancing Reliability and Availability

a. Inspection Scope

The team reviewed BECo's program for balancing availability and reliability and BECo's 1998 (a) (3) assessment report (issued in April 1998).

b. Observations and Findings

The team determined that BECo's program adequately implemented balancing availability and reliability. The balancing methodology was described in Procedure NE 16.03, "10 CFR 50.65 NRC Maintenance Rule," Revision 1, April 11, 1998. The method was essentially based on the premise that an appropriate balance between SSC reliability and availability is maintained when the performance of an SSC conforms to both its functional failure and unavailability performance criteria, and those criteria have been established with consideration of safety. In effect, the method was to verify that the performance criteria for availability and reliability were being met. This was a continuing process occurring through the day to day implementation of the Maintenance Rule.

The team reviewed BECO's (a)(3) assessment report and determined the report was appropriately self-critical ("Boston Edison Company - Pilgrim Nuclear Power Station - Maintenance Rule (a)(3) Periodic Assessment - July 10, 1996 through April 17, 1998 Operating Cycle 11," dated April 21, 1998). However, the assessment report was not completed in a timely manner and four sections (8.0 - Preventative Maintenance Optimization, 9.0 - Operating Experience Review, 11 - Plant walkdown, and 14 - Review of Documentation) of the report were not completed until the week after the inspection completed. The requirement for completion of the self assessment is linked to the refueling cycle. Refueling outage (RFO-11) was completed in April 1997 and therefore the (a)(3) assessment report should have been completed at the end of RFO 11. This is an apparent violation of 10 CFR 50.65 (a)(3). (EEI 50-293/98-04-05)

c. Conclusions

That BECo's program adequately implemented balancing availability and reliability. BECo's failure to complete the (a)(3) assessment at the end of RFO 11 in April 1997 is an apparent violation of 10 CFR 50.65 (a)(3).

M1.5 Plant Safety Assessments before Taking Equipment out of Service

a. Scope

The team reviewed BECo's conduct of on-line maintenance program with regard to its impact on the Maintenance Rule and risk assessment. In addition, the team interviewed a number of maintenance, engineering and operations personnel to ascertain their level of knowledge of the Rule and the relation between management of risk of on-line maintenance activities and equipment availability as it related to system performance. A review of risk associated with outage work activities was also performed.

b. Observations and Findings

The on-line maintenance program was described in procedure 1.5.21, "Work Control Scheduling Activities and Guidelines," Revision 2 and procedure 1.2.2, "Administrative Ops Requirements," Revision 16. On-line maintenance is planned through a rotating 12-week maintenance cycle (matrix). The matrix listed various combinations of equipment that could be taken out of service during the scheduled work week. BECo had evaluated the risk associated with these combinations of equipment out of service using probabilistic risk assessment (PRA). As of December 1997, BECo made the equipment out of service (EOOS) database available to assess the cumulative risk impact associated with taking equipment out of service. For the scheduled work activities, the team verified that the EOOS database was being used to assess the risk to the plant.

The EOOS program had the capability to generate an historical risk profile for the plant. The historical profile was the cumulative risk based on the risk levels of the various combinations of equipment and the duration the combination of equipment remained out of service. However, station personnel had not yet recognized its usefulness in implementing the maintenance rule. BECo's (a)(3) Periodic Assessment Report, Section 6.0, "Review of the Removal of Equipment from Service," identified that not all sources of risk are identified, not all risk significant SSCs are included, and there was no formal process for consideration of risk from emergent work. The report also noted that EOOS did not address reactivity monitoring and control, primary containment, external initiators such as weather, and internal initiators such as other work in the area, the condition of the redundant train if applicable, and the impact on the operators of all the activities being performed.

The team also noted that there was no mechanism to account for high risk significant SSCs into EOOS which were also not modeled in the PRA and had been upgraded by the expert panel. The team concluded this was an area of potential vulnerability.

A review of past logs revealed that on March 12, 1998, operators tagged out the high pressure coolant injection (HPCI) system for scheduled maintenance. Shortly thereafter, operators identified a problem with the station blackout (SBO) diesel and declared it inoperable. The team questioned BECo regarding whether the operating crew evaluated the increased risk or safety impact with both of these components out of service, and of the need to return the HPCI system back to service if the risk was determined to be unacceptable. The Assistant Operations Department Manager stated that the Nuclear Watch Engineer (NWE) did a qualitative assessment of the risk increase based on his training and experience as a licensed reactor operator, and concluded it to be acceptable. The NWE on watch at that time had not yet received the EOOS training. The team questioned the PRA engineer on what was the increase in risk with both the HPCI and SBO diesel out of service and was informed that it was minimal. The risk increase, based on the EOOS model, went from a risk factor of 8.2 to 8.8. This increase in risk is within that allowed by BECo. There was presently no procedural guidance associated with assessing the risk to the plant for emergent/work activities. The licensee indicated that all NWE's have since been trained on PRA and that they plan on using EOOS as a tool available for operators to use when assessing the risk impact for emergent work activities.

The team reviewed BECo's process for assessing risk for work activities during refueling outages. Risk assessment, while shutdown, was managed through the PNPS Outage Manual. Risk was managed through the use of a key safety functions process. The evaluation of risk used aspects of the at power PRA model (i.e. system dependencies) and the qualitative evaluation of risk based on engineering and operational judgment. BECo had recently developed an outage specific risk assessment model which would be used for the upcoming (cycle 12) refueling outage. A review of the cycle 11 refueling outage schedule and the compensatory measures established indicate that risk assessment was properly implemented.

c. Conclusion

The process for assessing risk associated with scheduled maintenance work activities was generally good and was being properly implemented at the site. Probabilistic risk assessments were not used to assess the overall plant risk when responding to emergent work activities during operations and when considering high risk significant SSCs which were not modeled in the PRA and had been upgraded by the expert panel. The persons interviewed expressed sufficient knowledge of the risk assessment process to implement the maintenance rule program. BECo's approach to shutdown risk program was reasonable.

M2 Engineering Support of Facilities and UFSAR Commitment Review

A discovery of a licensee operating their facility in a manner contrary to the Final Safety Analysis Report (FSAR) description highlighted the need for a special focused review that compared plant practices, procedures, and parameters to the FSAR descriptions. While performing the inspection discussed in this report, the team reviewed selected portions of the FSAR.

During the vertical slice reviews, the team noted one problem where information in the FSAR was not current. Specifically, the control room habitability analysis contained in Chapter 14.11 "Evaluation Using Standard NRC Approach" of the Pilgrim FSAR, did not contain the results of an updated, (circa 1982) control room habitability analysis performed in response to the requirements of NUREG 0737, "Clarification of TMI Action Plan Requirements" Item III.D.3.4, "Control Room Habitability" and as required by 10 CFR 50.71(e). BECo was aware of the need to update this chapter of the FSAR, and had formed a team to identify the out-of-date FSAR information, and devise a schedule to remove the incorrect information. The inspector noted the updated analysis did not appear to invalidate the conclusions reached by the earlier analysis. Therefore, this problem was of low safety significance and was viewed as a minor violation.

M3 Staff Knowledge and Performance

a. Inspection Scope

The team interviewed managers, system engineers, and operations department personnel to assess their understanding of the maintenance rule and their associated responsibilities.

b. Observations and Findings

The team determined the SE's knowledge of their systems was good. Their knowledge of the MR and the implementation of the rule was acceptable.

SROs and ROs had a basic awareness of the MR requirements and their responsibilities. They were less aware of PRA and risk bases for the MR. They knew where some of the key information could be found and who they could contact when questions arose. A limited amount of training had been recently provided to the licensed operators.

c. Conclusion

The system engineer and operations department personnel knowledge of the MR and their associated responsibilities were adequate to ensure acceptable implementation of the maintenance rule.

M7 Quality Assurance (QA) Related to Maintenance Activities**M7.1 Self-Assessments of the Maintenance Rule Program****a. Inspection Scope**

The team reviewed recent assessments related to the maintenance rule in order to determine if the provisions of the rule were properly implemented.

b. Observations and Findings

The QA self assessment, issued in October 1997, was a critical retrospective look at BECo's implementation of the MR at Pilgrim. Assessment findings were appropriately dispositioned. The October 1997 QA audit identified significant performance weaknesses and inadequacies which indicated an apparent organizational ineffectiveness prior to October, 1997 in implementing the maintenance rule. The team determined that the correct implementation of the maintenance rule program at the Pilgrim facility was due, in part, to their responsiveness to the audit findings. Although, BECo had implemented the maintenance rule acceptably at the time of the inspection with the exception of the violations identified, BECo did not meet the NRC expectation to have the maintenance rule fully implemented on or before July 10, 1996.

c. Conclusions

The team noted a number of revisions to the program and substantial improvements were implemented just prior to this inspection due in part to the thoroughness of the licensee's self-assessment and audit processes.

V. Management Meetings**X1 Exit Meeting Summary**

The team discussed the progress of the inspection with BECo representatives on a daily basis and presented the inspection results to members of management at the conclusion of the inspection on May 1, 1998.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

K. Burke, Operating Experience Coordinator
 D. Hanley, Maintenance Rule Coordinator, Expert Panel Chairman, PRA Data Analyst
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 M. Jacobs, Acting Operations Manager and Expert Panel Member
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 L. Wetherell, Nuclear Engineering Services Group Deputy Manager

NRC

R. Laura, Senior Resident Inspector

LIST OF INSPECTION PROCEDURES

IP 62706: Maintenance Rule

ITEMS OPENED AND CLOSED

Opened

VIO 50-293/98-04-01	EEI	Contrary to 10 CFR 50.65(a)(2), the licensee failed to establish unavailability performance criteria for the anticipated transient without scram (ATWS) function of the control rod drive system.
VIO 50-293/98-04-02	EEI	Contrary to 10 CFR 50.65 (a)(2), the licensee failed to place the feedwater system into an (a)(1) status in a timely manner.
VIO 50-293/98-04-03	EEI	Contrary to 10 CFR 50.65 (a)(2), the licensee failed to place the primary containment system into an (a)(1) status in a timely manner.

VIO 50-293/98-04-04	EEI	Contrary to 10 CFR 50.65(b) the licensee failed to include the reactor building 480V switch gear HVAC, and the firewater system function of providing a backup supply to the screenwash system, in the scope of the Maintenance Rule program.
VIO 50-293/98-04-05	EEI	Contrary to 10 CFR 50.65(a)(3), the licensee had failed to complete the (a)(3) evaluation within the required time interval.

DOCUMENTS REVIEWED OR REFERENCED

- NUMARC 93-01, "Nuclear Energy Institute - Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants [Line-In/Line-Out Version]," Revision 2, April 1996.
- Nuclear Organization Procedure NOP95A4, "NRC Maintenance Rule Procedure," Revision 1, April 11, 1998
- Nuclear Engineering Services Group Procedure 16.03, "10 CFR 50.65 NRC Maintenance Rule," Revision 1, April 10, 1998
- Pilgrim Nuclear Power Station, Individual Plant Examination (IPE) of Severe Accident Vulnerabilities in response to NRC Generic Letter 88-20 (Volumes 1, 2, and 3), submitted to the NRC by letter dated September 30, 1992.
- Pilgrim Nuclear Power Station, Response to NRC Request for Additional Information (RAI) concerning IPE, submitted to the NRC by letter dated December 28, 1995.
- Pilgrim Nuclear Power Station, Maintenance Rule SSC Basis Documents, all systems in the scope of the Maintenance Rule, April 1998.
- Expert Panel Meeting Minutes, February to April 1998.
- "Boston Edison Company - Pilgrim Nuclear Power Station - Maintenance Rule (a)(3) Periodic Assessment - July 10, 1996 through April 17, 1998 Operating Cycle 11".
- Design Review Board Meeting Minutes, March to April 1998.
- Pilgrim Nuclear Power Station Procedure 1.5.21, "Work Control Scheduling Activities and Guidelines," Revision 2, December 31, 1997.

- Pilgrim Nuclear Power Station Procedure 1.2.2, "Administrative Ops Requirements," Revision 16, November 15, 1996.
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- (a)(1) SSCs - Corrective Action Plans - Latest Revisions

LIST OF ACRONYMS

ADS - Automatic Depressurization System
 ATWS - Anticipated Transient Without Scram
 BECo - Boston Edison Company
 CAFTA - Computer Assisted Fault Tree Application
 CDF - Core Damage Frequency
 CM - Corrective Maintenance
 CRD - Control Rod Drive
 EP - Expert Panel
 EOPs - Emergency Operating Procedures
 EOOS - Equipment Out Of Service
 EPRI - Electric Power Research Institute
 FF - Functional Failure
 FSAR - Final Safety Analysis Report
 FV - Fussell-Vesely
 HPCI - High Pressure Coolant Injection
 IPE - Individual Plant Examination
 IPEEE - Individual Plant External Events Examination
 KV - Kilovolts
 LCO - Limiting Condition of Operation
 LERFs - Large Early Release Fractions
 LOCA - Loss of Coolant Accident
 MR - Maintenance Rule
 MRC - Maintenance Rule Coordinator
 MCC - Motor Control Center
 MOV - Motor Operated Valve
 MPFF - Maintenance Preventable Functional Failure
 NUMARC - Nuclear Management and Resources Council
 PASS - Post Accident Sampling System
 PC - Performance Criteria
 PRA - Probabilistic Risk Assessment
 QSS - Quarterly System Schedule
 RAW - Risk Achievement Worth
 RCIC - Reactor Core Isolation Cooling

RCS - Reactor Coolant System
RHR - Residual Heat Removal System
RPS - Reactor Protection System
RRW - Risk Reduction Worth
SBO - Station Blackout
SEP - Safety Enhancement Program
SSCs - Structures, Systems, and Components